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April 30, 1982

NUCLEAR PRODUCTION DEPARTMENT



U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Units 1 and 2
Docket Nos. 50-416 and 50-417
File: 0272/M001.0
FSAR and Technical Specification Changes Pertaining to RPS Circuit Response Times
AECM-82/142

This transmittal is provided to inform the NRC of proposed changes to the GGNS FSAR and Technical Specifications as a result of actual delay response time measurements performed on the Turbine Stop and Control Valve Closure circuits and the RPV Water Level Transmitter circuits.

During response time testing several of the main turbine stop and control valves circuits failed to meet the 70 msec. total maximum response time specified in the GGNS Technical Specifications (Table 3.3.1-2). During similar testing, the RPV water level transmitters circuits failed to meet the Technical Specification total response times of 300 msec.

As a result of the higher measured total response times, the GGNS Technical Specification limits require revision in order that the limiting conditions for operation can be established for these circuits. The specification for total response time for the turbine stop and control valves (presently 60 and 70 msec., respectively) has been increased to 100 msec. for both circuits. The RPV water level transmitter total response time has been increased from 300 msec. to 1.05 seconds. These changes have been provided to Mr. Bottimore of the NRC and have been incorporated into the proof and review copy of the GGNS Technical Specifications. The revised proof and review copy of Table 3.3.1-2 is provided in Attachment 1.

The primary effect to be experienced from the longer response times will be on the transient analyses performed for Grand Gulf, due to the additional delay time prior to initiation of reactor scram.

The most limiting transient for Grand Gulf, which is the Loss of Feedwater Heating (Manual Flow Control), is controlled by the high flux initiated scram and is, therefore, uneffected by either response time change being proposed.

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The limiting transients which are effected by a turbine control and stop valve scram trip signal are the Generator Load Rejection without Bypass, Generator Load Rejection with Bypass, and the Turbine Trip without Bypass. Each of these transients, which are pressurization transients, were reanalyzed using the ODYN Code to determine their effect on reduction in the minimum critical power ratios (MCPR). The Feedwater Controller Failure at Maximum Demand is the limiting transient effected by a longer RPV water level transmitter response time. Likewise, a reanalysis of this transient was performed with the longer total response time of 1.05 seconds. In each case, no significant change in CPR's was observed, and the MCPR's as reported in FSAR Table 15.0-1 (Amendment 55, April, 1982) are accurate.

Therefore, for the GGNS transient analyses, it has been determined that no effect on the ability to safely operate the plant will be experienced from a change in these response times.

In order to avoid potential misinterpretation between the FSAR and the Technical Specifications, FSAR Table 7.2-5 is being deleted as it presently exists and the total response times, used as transient analysis input parameters, will be included in the response to NRC Question 211.134. The proposed FSAR revisions are provided in Attachment 2. The incorporation of these proposed revisions into the FSAR will be made pending the receipt of further guidance from the NRC in regard to post-operating license FSAR amendments.

Please advise if any additional information is required.

Yours truly,

John J Richardson for

L. F. Dale
Manager of Nuclear Services

SAB/JGC/JDR:rg

Attachments

cc: Mr. N. L. Stampley (w/a)
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TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

| FUNCTIONAL UNIT | RESPONSE TIME (Seconds) |
|---|---|
| 1. Intermediate Range Monitors: | |
| a. Neutron Flux - High | NA |
| b. Inoperative | NA |
| 2. Average Power Range Monitor*: | |
| a. Neutron Flux - High, Setdown | NA |
| b. Flow Biased Simulated Thermal Power - High | ≤ 0.09** |
| c. Neutron Flux - High | ≤ 0.09 |
| d. Inoperative | NA |
| 3. Reactor Vessel Steam Dome Pressure - High | ≤ 0.35 |
| 4. Reactor Vessel Water Level - Low, Level 3 | ≤ 1.05 |
| 5. Reactor Vessel Water Level - High, Level 8 | ≤ 1.05 |
| 6. Main Steam Line Isolation Valve - Closure | ≤ 0.06 |
| 7. Main Steam Line Radiation - High | NA |
| 8. Drywell Pressure - High | NA |
| 9. Scram Discharge Volume Water Level - High | NA |
| 10. Turbine Stop Valve - Closure | ≤ 0.10 |
| 11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low | ≤ 0.10# |
| 12. Reactor Mode Switch Shutdown Position | NA |
| 13. Manual Scram | NA |

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.
**Not including simulated thermal power time constant.
#Measured from start of turbine control valve fast closure.



Attachment 1
AECM-82/142

211.134 (15.0) The analysis of transients and accidents in Chapter 15.0 does not state which of the RPS time response delays in Table 7.2-5 is used in the REDY computer model (NEDO-10802). For each transient and accident in Chapter 15.0, specify which delay time in Table 7.2-5 is used in the analysis and why specified delay time is conservative.

RESPONSE

The total RPS response time used in both REDY and ODYN transient analysis codes are provided below.

| Function | Total Response Times (Seconds) |
|---|---|
| IRM neutron flux[1] | 0.11 |
| APRM neutron flux | <0.09 |
| Reactor vessel high pressure | 0.35 |
| Reactor vessel low water level | 0.30 (1.05)[3] |
| MSLIV closure | 0.06 |
| MSL high radiation | 1.05 |
| Drywell high pressure | 0.65 |
| Scram discharge volume high water level | 1.05 |
| Turbine stop valve closure | 0.07 (0.10)[2] |
| Turbine control valve fast closure | 0.07 (0.10)[2] |
| Reactor vessel high water level | 0.30 (1.05)[3] |

(1) Time delay requirements are applicable only above 0.4 percent or rated power.

(2) The total response time indicated is based on testing data as discussed in AECM-82/142. The generator load rejection with bypass. generator load rejection without bypass, and turbine trip without bypass transients were reanalyzed using this response time with no significant effect on Δ CPR's.

(3) The total response time indicated is based on testing data as discussed in AECM-82/142. The feedwater controller failure (maximum demand) transient was reanalyzed using this response time with no significant effect on the Δ CPR's.

The fourth test is the single rod scram test, which verifies the capability of each rod to scram. It is accomplished by operating two toggle switches on the hydraulic control unit for the particular control rod drive. Timing traces can be made for each rod scrammed. Prior to the test, a physics review must be conducted to assure that the rod pattern during scram testing will not create a rod of excessive reactivity worth.

The fifth test involves applying a test signal to each reactor protection system sensor trip channel, in turn, and observing that a logic trip results. This test also verifies the electrical independence of the channel circuitry. The test signals can be applied to the process type sensing instruments (pressure and differential pressure) through calibration taps. Calibration and test controls for pressure transmitters, level transmitters, and valve position switches are located in the turbine building and containment. To gain access to the setting controls on each transmitter, a cover plate or sealing device must be removed. The control room operator is responsible for granting access to the setting controls. Only properly qualified plant personnel are granted access for the purpose of testing or calibration adjustments. 031.48

Transmitter operation will be ascertained during plant operation by comparison of the four individual channel trip units. Any deviation of a reading from the norm (other units) would indicate a malfunction.

Transmitter testing and calibration will normally take place during plant outage. 031.57 a

The alarm typewriter provided with the process computer shows verification of the correct operation of many sensors during plant startup and shutdown. Main steam line isolation valve position and turbine stop valve position can be checked in this manner. The verification provided on the alarm typewriter is not considered in the selection of test and calibration frequencies and is not required for plant safety. 37

The overall reactor protection system response time is verified during preoperational testing from sensor trip to sensor trip channel relay de-energization and actuator de-energization, and can be verified thereafter by similar testing. RPS total response time values are provided in Table 3.3.1-2 of the GGNS Technical Specifications. 031.48

7.2.1.1.5 Environmental Considerations

Electrical modules for the reactor protection system are located in the drywell, containment, and the turbine building. The environmental conditions for these areas are shown in Table 3.11-1.

TABLE 7.2-5

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